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Title: INITIAL UC Davis Reactor Analysis Context

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## INITIAL UC Davis Reactor Analysis Context

MNRC Safeguards, Core Cooling, and Calorimetric laboratory - 3.0 hours.

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## 1 Introduction

The INITIAL reactor experiment pitched to UC Davis and agreed upon by the professor teaching the course is a natural uranium neutron activation experiment in the McClellan Nuclear Research Center (MNRC) reactor. Both non-cadmium covered (Ex:1 – bare) and cadmium covered (Ex:2 – Cd) experiments will be performed and analysis will ensure on gamma spectra collected from both experiments. The following provides necessary information for completing the experiment analysis. This information includes definitions, assumptions, and plots tailored to the specifics of this experiment. The content is for the students to be completing the laboratory for reference.

## 2 Activation Constraints/Conditions

Material	– The material to be irradiated is natural uranium dioxide.
	– Chemical Form - $\text{UO}_2$
	– Density - 10.97 g/cc
	– Enrichment 0.72 at.% $^{235}\text{U}$
	– Mass - 10 grams
Irradiation Time	– Unless otherwise specified:
	1. Bare Experiment: 4.6 hours
	2. Cd experiment: 24 hours
Cooling Time	– Unless otherwise specified:
	1. Bare Experiment: $\sim 30$ days
	2. Cd experiment: $\sim 30$ days
Neutron Scalar Flux	– Unless otherwise specified:
	1. Bare Experiment: $\sim 8.75 \times 10^{11} \text{ n}/(\text{cm}^2 \cdot \text{s})$
	2. Cd experiment: $\sim 5.75 \times 10^{11} \text{ n}/(\text{cm}^2 \cdot \text{s})$

### 3 Analysis Background

#### 3.1 Time Since Removal from Reactor

To verify the operators declaration of time since removal from the reactor, Figures 1 and 2 are provided. These figures show the absolute fraction of activity due to different isotopes in the system for both experiments. When determining relative activities, gamma peaks shown in Table 1 should be used. These peaks are chosen so that interferences are minimal<sup>i</sup>

If the professor is nice enough to declare a cooling time for the fuel, then Figures 1 and 2 are used for verification. This is done by determining relative activities<sup>ii</sup> for isotopes that can be seen in the gamma spectrum.

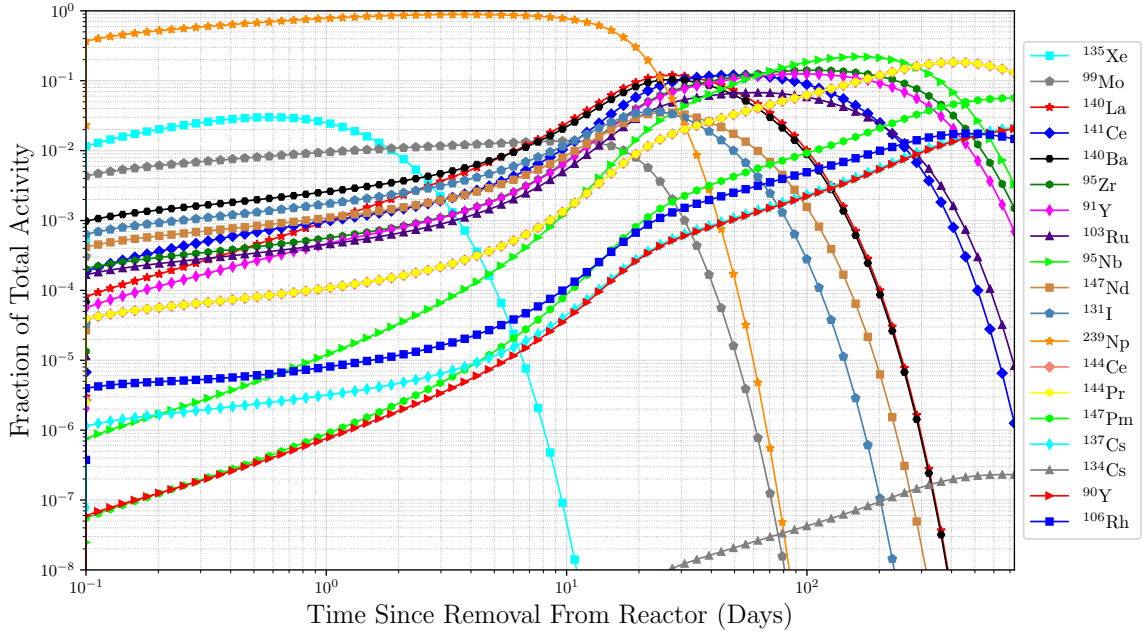


Figure 1: Activity fractions as a function of time since removal from reactor as determined with ORIGEN (**Bare Experiment**).

If a cooling time is not declared, then it is suggested to first determine the relative activities<sup>iii</sup> of  $^{239}\text{Np}$  and another isotope that is increasing (ex.  $^{144}\text{Pr}$ ,  $^{137}\text{Cs}$ , or  $^{106}\text{Rh}$ ) and determine whether cooling time is greater than or less than their Cross Over Time (COT, time when activities are equal, ex. COT for  $^{239}\text{Np}$  and  $^{144}\text{Pr}$  is near 30 days (Bare Experiment) and 40 days (Cd Experiment))<sup>iv</sup>. Once the region of decay time (either greater or less than COT) is established, then more isotopes can be brought into the mix to narrow down exactly when the material was removed from the reactor.

<sup>i</sup>An example interference is  $^{147}\text{Pm}$ 's largest yield gamma peak at 121 keV. This peak is not included in Table 1 because it overlaps with  $^{147}\text{Nd}$  120 keV peak, and  $^{140}\text{Ba}$  119 keV peak.  $^{144}\text{Ce}$  is not included in this list because a number of its peaks interferes with higher activity isotopes (Note: Made interference calculations at 30 days decay (wrote an awesome code to do this)).

<sup>ii</sup>Described in the appendix

<sup>iii</sup>Again, in the appendix

<sup>iv</sup>Hint: If  $A_{^{239}\text{Np}} > A_{^{144}\text{Pr}}$  then  $T < \text{COT}$ . If you can't see  $^{239}\text{Np}$ , this is a good indicator that  $T > \text{COT}$ .

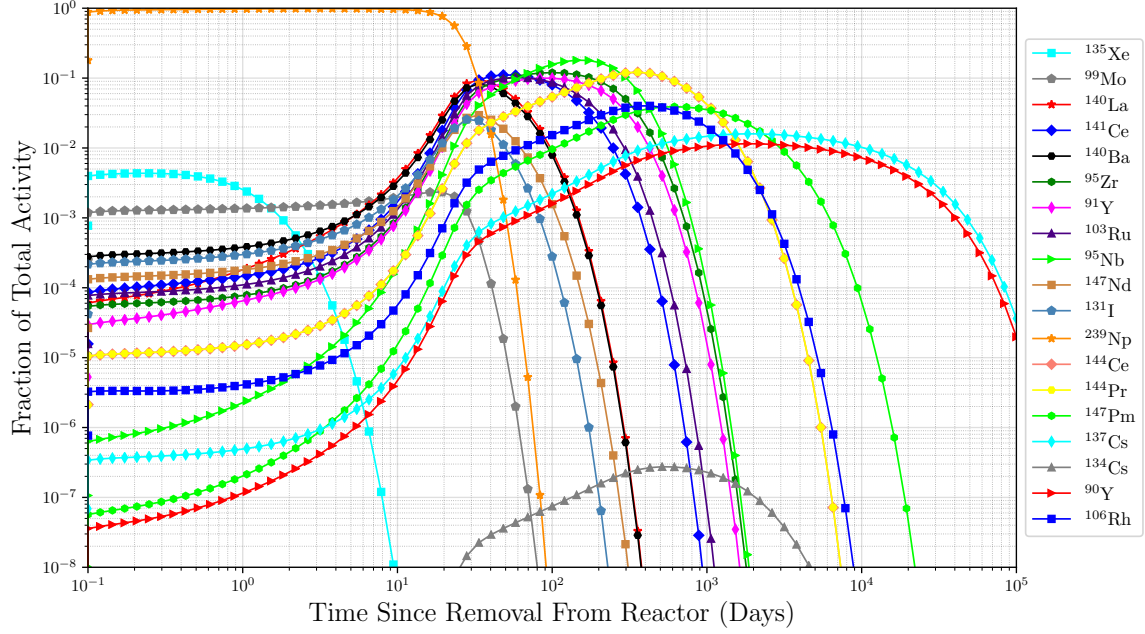


Figure 2: Activity fractions as a function of time since removal from reactor as determined with ORIGEN (Cd Experiment).

Table 1: Gamma energy peaks to use for time since removal from reactor verification.

Isotope	$\lambda$ ( $s^{-1}$ )	Energy[keV](Yield[%])
$^{140}\text{La}$	$4.78 \times 10^{-06}$	1596(95.4), 487(45.5), 816(23.3)
$^{143}\text{Pr}$	$5.91 \times 10^{-07}$	742( $1.2 \times 10^{-06}$ )
$^{141}\text{Ce}$	$2.47 \times 10^{-07}$	145(48.2)
$^{140}\text{Ba}$	$6.29 \times 10^{-07}$	537(24.4), 30(14.1), 163(6.2)
$^{95}\text{Zr}$	$1.25 \times 10^{-07}$	757(54.0), 724(44.2), 236(0.3)
$^{91}\text{Y}$	$1.37 \times 10^{-07}$	1205(0.3)
$^{103}\text{Ru}$	$2.04 \times 10^{-07}$	497(90.9), 610(5.8), 444(3.3)
$^{95}\text{Nb}$	$2.29 \times 10^{-07}$	766(100.0), 204( $2.8 \times 10^{-02}$ )
$^{147}\text{Nd}$	$7.31 \times 10^{-07}$	91(28.0), 531(13.1)
$^{131}\text{I}$	$1.00 \times 10^{-06}$	364(81.7), 637(7.2)
$^{239}\text{Np}$	$3.40 \times 10^{-06}$	106(27.2)
$^{144}\text{Pr}$	$6.69 \times 10^{-04}$	697(1.3), 2186(0.7), 1489(0.3)
$^{132}\text{I}$	$8.39 \times 10^{-05}$	773(75.6), 955(17.6)
$^{147}\text{Pm}$	$8.38 \times 10^{-09}$	76( $1.2 \times 10^{-08}$ )

### 3.2 Production of $^{137}\text{Cs}$

Several samples of natural uranium (0.72 at.%  $^{235}\text{U}$ ) are to be irradiated in the neutron flux spectra of the MNRC. In the course of irradiation the composition of the fuel will change due to fissioning of fuel into fission products (FP), transmutation of nuclides from neutron absorption to create TRansUranics (TRU – elements with more protons than uranium), and radioactive decay (among other things). The concentration as a function of time of each component in the fuel (U, TRU, FP) can be described with a large system of differential equations. For example, the concentration of  $^{235}\text{U}$  in the system can be described under the 1-group **assumption**<sup>v</sup> with Equation 1.

$$\frac{\delta N_{235}(t)}{\delta t} = -(\lambda_{235} + \phi\sigma_{235})N_{235}(t) \quad (1)$$

where:

$N_{235}$  = the number of  $^{235}\text{U}$  atoms in the presence of the scalar flux.

$\lambda_{235}$  = the radiological half-life of  $^{235}\text{U}$

$\phi$  = the total scalar flux and

$\sigma_{235}$  = the average<sup>vi</sup> total transmutative cross section for  $^{235}\text{U}$  (absorption, fission, (n,2n), etc.)

If we **assume**  $\phi$  and  $\sigma$  are constant as a function of time, then the solution to Equation 1 is a simple exponential.

$$N_{235}(t) = N_{0,235}e^{-(\lambda_{235} + \phi\sigma_{235})t} \quad (2)$$

As a matter of fact, the solution to every isotope in the system is an exponential of one form or another. In the case of Equation 2, if we **assume** short irradiation times, then  $N_{235}(t) \approx N_{0,235}$ . The assumption that the number of  $^{235}\text{U}$  atoms in the system is constant will help with the derivation for the concentration of one of the fission products.

In the case where only  $^{235}\text{U}$  is undergoing fission, the differential equation for the concentration of  $^{137}\text{Cs}$  could be written as shown in Equation 3.

$$\frac{\delta N_{137}(t)}{\delta t} = -(\lambda_{137} + \sigma_{137}\phi)N_{137}(t) + \gamma_{137}N_{0,235}\phi\sigma_{f,235} \quad (3)$$

where:

$\gamma_{137}$  = the cumulative fission yield for  $^{137}\text{Cs}$   $\left[ \frac{\# \text{ of } ^{137}\text{Cs atoms}}{\text{fission of } ^{235}\text{U}} \right]$ <sup>vii</sup>  $\approx 0.0627$

$\sigma_{137}$  = the average total transmutative cross section for  $^{137}\text{Cs} \approx 0$

$\sigma_{f,235}$  = the average fission cross section for  $^{235}\text{U}$

Note: **Assuming** for  $\gamma_{137}$  that all fission isotopes in the 137 mass bin immediately arrive at  $^{137}\text{Cs}$  without passing through any of the precursors (precursor examples include:  $^{137}\text{Xe}$ ,  $^{137}\text{I}$ ,  $^{137}\text{Te}$ ).

<sup>v</sup>Also **assuming** homogeneous flux and material. Additionally that the sample being irradiated does not perturb (or change) the flux spectrum.

<sup>vi</sup>In this context “average” refers to the flux averaged (over energy) cross section  $\sigma = \int \sigma(E)\phi(E)dE / \int \phi(E)dE$ .

<sup>vii</sup>Note: Will **assume** this is constant for all fissioning isotopes and energies of neutrons causing fission.

Given the assumptions above the solution for the number of  $^{137}\text{Cs}$  atoms as a function of time is given in Equation 4<sup>viii</sup>.

$$N_{137}(t) = \gamma_{137} N_{235} \phi \sigma_{f,235} t \quad (4)$$

### 3.3 Fuel Burn-up (Direct Calculation)

Fuel burn-up can be defined in several ways. One way is an operational parameter for the amount of energy produced per unit mass of the fissionable material. For a constant power, burnup is shown in Equation 5.

$$BU = \frac{\text{Power [MW]} \cdot \text{days}}{\text{mass [tHM]}} \quad (5)$$

where:

Power = thermal power released into the working fluid

days = number of days operated at Power

mass = the initial mass of heavy metal in the irradiated fuel in metric tons

In order to estimate burnup with fission cross sections and number densities of fissionable material, Equation 6 can be used (with units and values showed for clarity and for future calculations).

$$BU \left[ \frac{\text{MWd}}{\text{tHM}} \right] = \frac{C \left[ \frac{\text{MWd}}{\text{MeV}} \right] \cdot \phi_{tot} \left[ \frac{n}{\text{cm}^2 \cdot s} \right] \cdot T[s]}{m_{HM}[\text{tons}]} \cdot \sum_i N_i (\sigma_{f,i} E_f + \sigma_{\gamma,i} E_{\gamma}) \left[ \frac{\text{cm}^2 \cdot \text{MeV}}{n} \right] \quad (6)$$

where:

- $E$  = Energy released per fission (f) or capture ( $\gamma$ ) reaction
  1. Fission: **Assuming** 200 MeV/fission for all isotopes
  2. Capture:  $^{235}\text{U}$  – 6.5 MeV,  $^{238}\text{U}$  4.5 MeV
- $\sigma_{\gamma}$  = capture single-group neutron cross section
  1. Bare experiment:  $^{235}\text{U}$  – 30 b,  $^{238}\text{U}$  – 5.6 b
  2. Cd experiment:  $^{235}\text{U}$  – 4.3 b,  $^{238}\text{U}$  – 7.4 b
- $\sigma_f$  = fission single-group neutron cross section
  1. Bare experiment:  $^{235}\text{U}$  – 162 b,  $^{238}\text{U}$  – 0.076 b
  2. Cd experiment:  $^{235}\text{U}$  – 9.75 b,  $^{238}\text{U}$  – 0.12 b
- $\phi_{tot}$  = total scalar neutron flux
  1. Bare experiment:  $\phi_{therm}/f_{therm} = 3 \times 10^{11}/0.34$   
 $f_{therm} = 0.34$
  2. Cd experiment:  $\phi_{tot} \cdot (1 - f_{therm}) = 5.75 \times 10^{11} \text{ n}/(\text{cm}^2 \cdot \text{s})$
- $N_i$  = Atoms:  $^{235}\text{U}$  –  $1.61 \times 10^{20}$ ,  $^{238}\text{U}$  –  $2.2 \times 10^{22}$   
(10 grams of Nat.  $\text{UO}_2$ )

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<sup>viii</sup>Why am I ignoring the  $(\lambda_{137} + \sigma_{137}\phi)N_{137}(t)$  term on the RHS of Equation 3, and what two **assumptions** are included in this ignoration?

- $C$  = Energy conversion factor  $1.9 \times 10^{-24} \left[ \frac{\text{MWd}}{\text{MeV}} \right]$   
 $m_{HM}$  = the metric tons of heavy metal in the system ( $8.8 \times 10^{-6}$  tons for 10 grams of Nat.  $\text{UO}_2$ )  
 $i$  = index for fissionable isotopes in the system (Hint:  $i$  will equal  $^{235}\text{U}$  and  $^{238}\text{U}$  – we can ignore  $^{239}\text{Pu}$ )  
 $T$  = the irradiation time.

Note that many of the assumptions that applied to Equation 4 also apply to Equation 6. Also the connection between Equation 5 and Equation 6 can be made by understanding that the term,  $\phi N \sigma E$ , has units of MeV released per second.

We can determine burnup from our measurements without prior knowledge of the neutron flux spectrum. Equation 7 shows how this can be done<sup>ix</sup>.

$$\text{BU} = \frac{N_{137}}{N_{0,HM}} \cdot \frac{N_A E_f C}{\gamma_{137}} \left[ \frac{\text{MWd}}{\text{mols}} \right] \cdot \frac{1}{M_0^{HM}} \left[ \frac{\text{mols}}{\text{tHM}} \right] \quad (7)$$

where:

$N_A$  = Avogadro's constant

$M_0^{HM}$  = the initial heavy metal molar mass ( $\sim 2.38 \times 10^{-4}$  tHM/mol),

and all other terms have been previously defined. It should be noted that energy release from radiative capture is not included in Equation 7, where it is included in Equation 6. The effect this will have on burnup estimates is left to the student for explanation.

### 3.4 Fuel Burn-up (Typical Fission Product Ratio Calculation)

With large fuel assemblies, the absolute activity of  $^{137}\text{Cs}$  is difficult to determine due to self-shielding. Another method to determine the burnup of a fuel assembly is to use a ratio of activities, so that absolute amounts of a particular isotope is not necessary. Two common isotope ratios are  $^{134}\text{Cs}/^{137}\text{Cs}$  and  $^{154}\text{Eu}/^{137}\text{Cs}$ [1, Ch. 18 – Sec. 3.5]. These ratios are used because the relationship is linear as a function of burnup.

Figure 3 shows a plot of these two isotopes as a function of burnup for both our experiments and has been calculated using MCNP generated neutron flux spectra coupled with SCALE's ORIGEN module. It should be noted that the logarithmic scale on the y-axis makes the relationship look non-linear and it is also important to note that the slopes for the two different experiments are different. Differences in slope are due to differences in flux spectra (Bare  $\sim$  thermal, Cd  $\sim$  fast) leading to the use of different fission product yields in the calculation as well as different reactions being favorable.

A final point of note for Figure 3. There are two main production schemes for  $^{134}\text{Cs}$  and  $^{154}\text{Eu}$ . Direct fission product yield to the isotope (ex. Fission  $\rightarrow$   $^{134}\text{Cs}$ ) and neutron capture on a lighter nuclei (ex. Fission  $\rightarrow$   $^{133}\text{Cs} \rightarrow$   $^{134}\text{Cs}$ ). In power reactors, the second scheme reflects the majority of  $^{134}\text{Cs}$  and  $^{154}\text{Eu}$  production, whereas in our system, the majority comes from the first. The reason for this is because the total scalar neutron flux is much less in a research reactor (noting also that the second scheme is proportional to the total scalar neutron flux squared).

To determine the burnup with these plots one needs to fit a line to each curve with the origin as an intercept, and determine the appropriate activity ratio.

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<sup>ix</sup>Can you derive Equation 7 from Equations 6 and 4

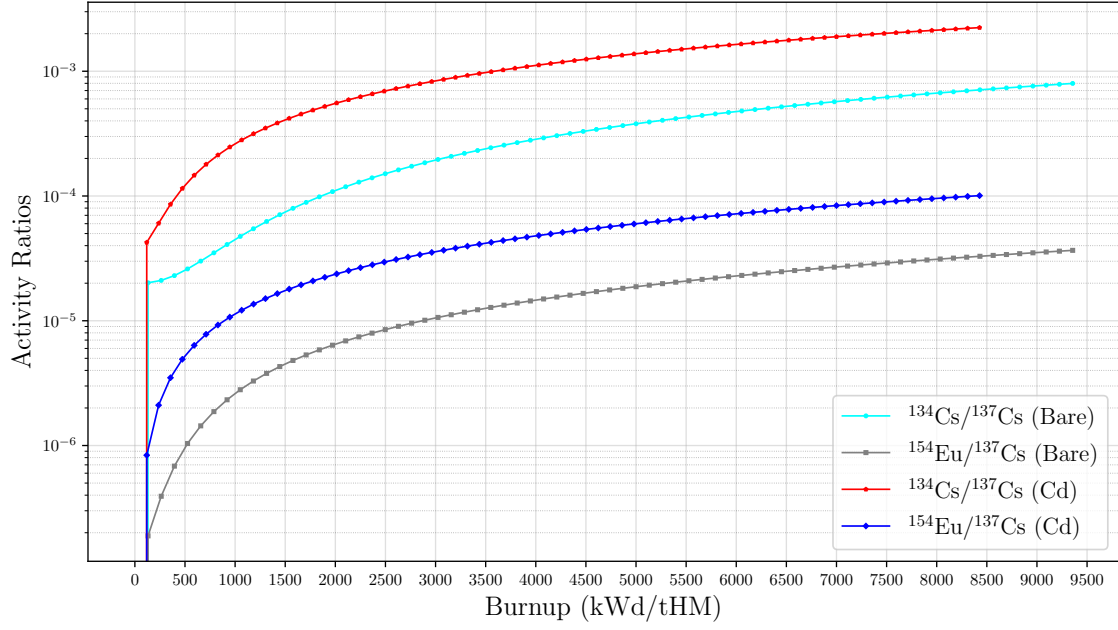


Figure 3:  $^{134}\text{Cs}/^{137}\text{Cs}$  and  $^{154}\text{Eu}/^{137}\text{Cs}$  activity ratios as a function of burnup as determined with ORIGEN.

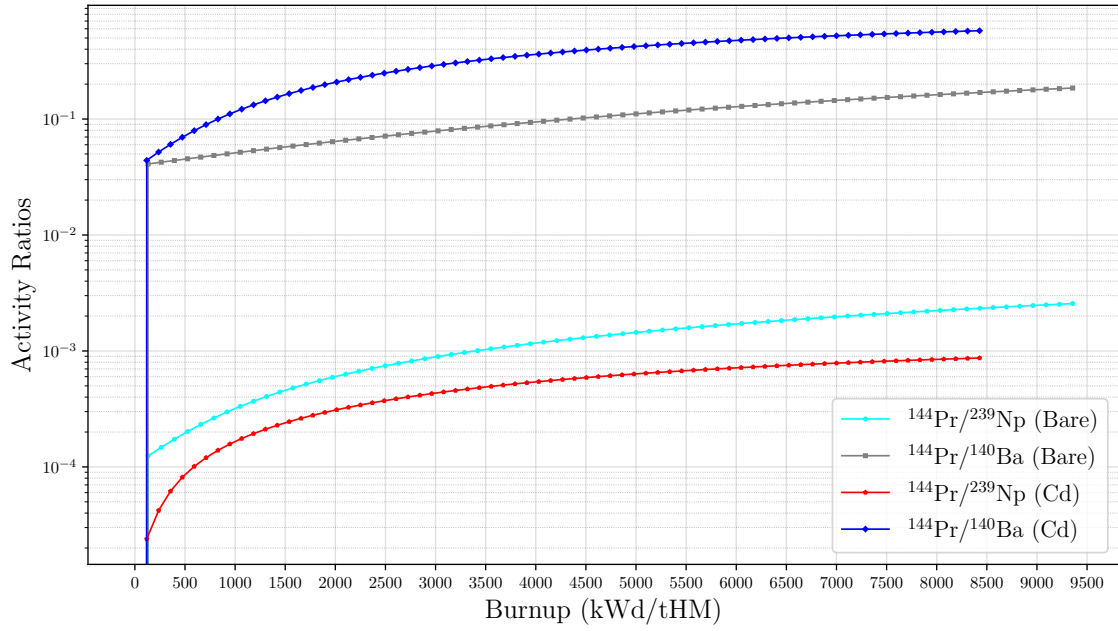


Figure 4: Atypical activity ratios as a function of burnup as determined with ORIGEN.

### 3.5 Fuel Burn-up (Atypical Fission Product Ratio Calculation)

In our situation, our irradiated  $\text{UO}_2$  might not have radioactively cooled long enough for the  $^{134}\text{Cs}$ ,  $^{154}\text{Eu}$ , or  $^{137}\text{Cs}$  to be distinguishable from all the other high activity fission products. Some ratio plots are provided in Figure 4 for some of the shorter lived isotope ratios as a function of burnup. **Please note: Proper<sup>x</sup> decay correction to the end of irradiation time is necessary for Figure 3 or Figure 4 to be beneficial<sup>xi</sup>.**

### 3.6 TRU concentrations in the fuel

Figure 5 shows how  $^{238}\text{U}$  is transmuted via neutron absorption ( $\sigma_\gamma$ ) and radioactive decay ( $\beta^-$ ) during irradiation to produce different plutonium isotopes. It could imagined how heavier isotopes and elements could be created in the system.

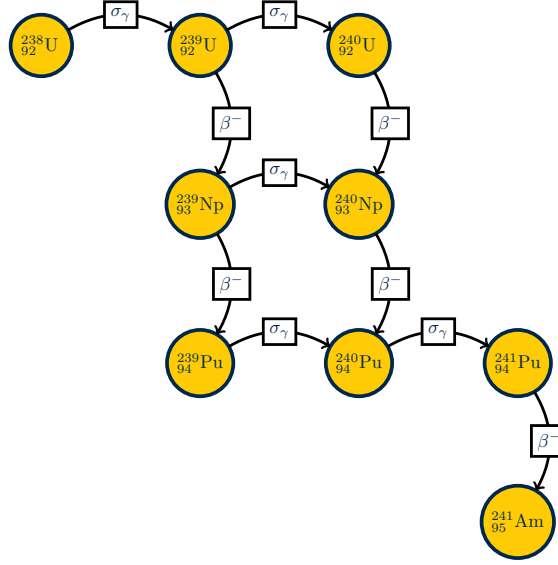


Figure 5: A transmutation network for  $^{238}\text{U}$ .

Describing such a system with a series of differential equations is possible, and analytical solutions exist for each isotope, but this can get involved. It should be known though, that as neutron fluence increases, the mass of the TRansUranics (TRU) will increase as uranium is converted to the heavier isotopes. These heavy isotopes are unstable and will mostly undergo alpha decay with small probabilities for spontaneous fission.

### 3.7 Decay Heat/Dose Calculations

Residual heat in irradiated fuel is due to radioactive decay. As described previously, fission products<sup>xii</sup> and TRansUranics<sup>xiii</sup> are produced during the course of irradiation. These heavy and “light” radioactive species emit different radiation. The longer lived heavier elements mostly decay by  $\alpha$  ( $^4_2\text{He}$ ) emission or through spontaneous fission. Both processes also emit photons. The fission products mostly decay by beta (a fast electron) emission. This process also emits

<sup>x</sup>Don’t decay correct the  $^{144}\text{Pr}$  (notice its *really* short half-life), but rather the  $^{144}\text{Ce}$  that feeds the  $^{144}\text{Pr}$ .

<sup>xi</sup>Hint: You might get better answers using the  $^{239}\text{Np}$  normalization if you can see its peak. If you can’t see the 106 keV peak for  $^{239}\text{Np}$  then use the early portion ( $<1000$  kWd/tHM) of the  $^{140}\text{Ba}$  normalization lines.

<sup>xii</sup>Example of  $^{137}\text{Cs}$  used in Section 3.2

<sup>xiii</sup>Example of Pu used in Section 3.6

photons. For a visual representative, the Bare experiment was simulated in ORIGEN, and the fraction of decay heat associated with these radiations has been plotted as a function and is shown in Figure 6.

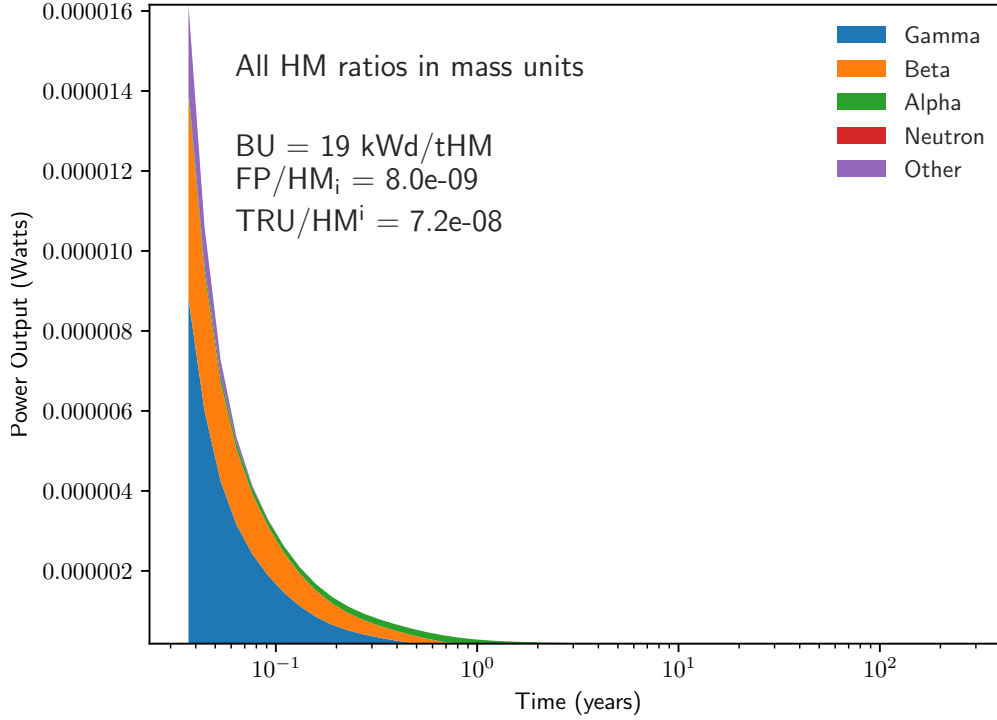


Figure 6: Decay heat plot for bare experiment.

Notice how Figure 6 shows that for our experiment, most of the decay heat radiation comes from the gamma emissions. The power output is also very small because of the small mass of fuel irradiated. It should be noted that all this decay heat will **not** deposit inside the fuel because gamma radiation has a good chance of escaping the material.

So that we do not get the wrong idea, Figure 7 is provided to show what the decay heat signature would look like if our bare experiment irradiation continued until the burnup was comparable to power reactor fuel burnup. It should be noted the differences in overall watt output, burnup, fission product (FP) mass and TRU mass in either scenario, as these might be helpful for answering laboratory questions.

To estimate the dose (or heating) from each of these species for a **short irradiation experiment**, it would be helpful to assume that alpha and neutron emissions are negligible. Further, once a time of decay has been estimated for both experiments, Figures 1 and 2 would be helpful for determining activities of any species that could not be determined with a gamma spectrum (one would need to determine the absolute activity of a single isotope and use that isotope, along with either Figure 1 or 2 – depending on experiment – to determine the activity of the other species in the fuel). Once the activity contributors have been determined, a useful tool for determining how much heat a particular isotope is emitting is an Evaluated Nuclear Data File (ENDF). Specifically, MF 8, MT 457. If those numbers do not make sense, then follow this link: <http://www.nndc.bnl.gov/exfor/endf00.jsp> and use Figures 8 – 10 in the appendix

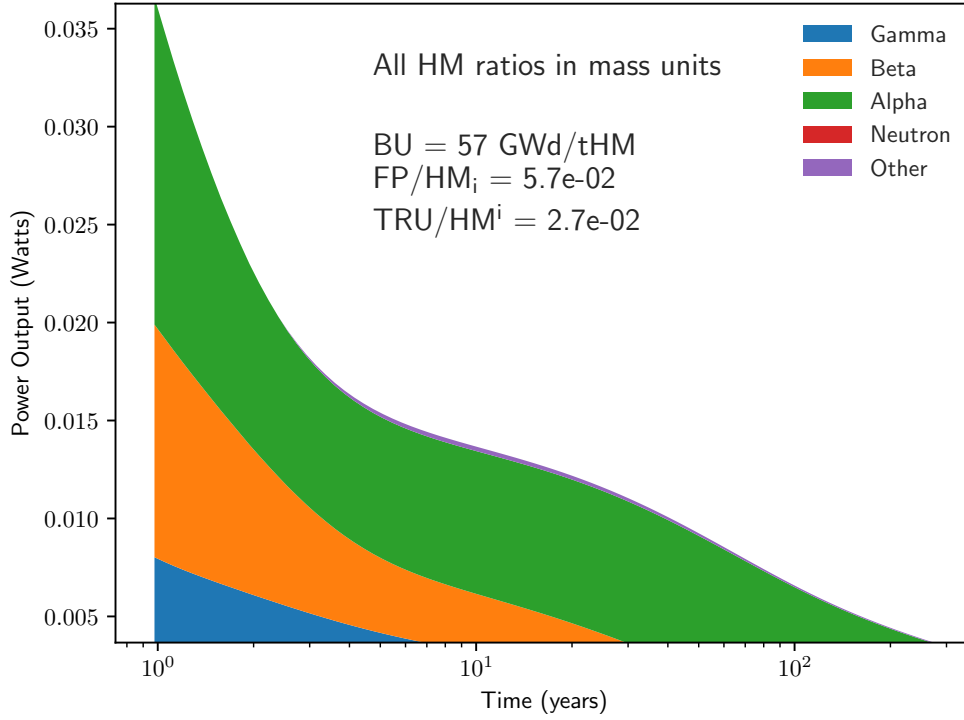


Figure 7: Decay heat plot for bare experiment.

for help in navigation<sup>xiv</sup>.

The **total** energy emission could be determined with each isotope using this information. Equation 8 shows how this would be done for an individual isotope. For gamma energies, the energies associated with gammas, x-rays, and 511 annihilations would need to be substituted for the  $Q - E_{\text{neutrino}}$  term.

$$\text{Heat} = A \cdot (Q - E_{\text{neutrino}}) \quad (8)$$

## References

- [1] DT Reilly, N Ensslin, HA Smith Jr, and S Kreiner. Passive nondestructive assay of nuclear materials (panda) manual. Technical report, NUREG/CR-5550 (US Government Printing Office, Washington, DC, 1991). Los Alamos National Laboratory document LA-UR-90-732, 1991.

<sup>xiv</sup>The example in the figures is  $^{60}\text{Co}$ , please substitute for your own isotopes of interest

## Appendix

### Relative Activities

Absolute activities will be needed for the decay heat calculation. For the time since removal calculation, a relative activity could be used. For example, using the 1205 keV peak for  $^{90}\text{Y}$ , and the 106 keV peak for  $^{239}\text{Np}$ . The relative activity for  $^{239}\text{Np}$ , among these two isotopes could be determined with Equation 9.

$$R_{A,239} = \frac{A_{239}}{A_{239} + A_{90}} \quad (9)$$

Where a single activity is determined with Equation 10. Note that if the geometric efficiency is not known, it does not matter because its value cancels out when plugged into Equation 9, only an energy calibration is needed.

$$A_{239} = \frac{CPS_{1205}}{BR_{1205} \epsilon_{Energy=1205} \epsilon_{geo}} \quad (10)$$

In order to use this information to determine the time since removal from reactor, find the fraction of activity of these two isotopes at your suspected time (use Figure 1 or 2), and add use those fractions,  $f_{239}$  and  $f_{90}$ , to determine another relative activity, shown in Equation 11. If the relative activities determined from Figure 1 or 2 match with the experimentally determined value, then the suspected time since removal from reactor is correct. If not, then another time should be subject to the same methodology. The shape of the graphs might be able to inform what the next guess of time since removal from reactor should be. This method can be expanded to include as many isotopes as measurable.

$$R_{A,239} = \frac{f_{239}}{f_{239} + f_{90}} \quad (11)$$

As a potentially helpful aside for students, if an absolute activity for a single species were known along with the time since removal from reactor. Then the activity of another species in the system could be estimated with Equation 12.

$$A_2 = A_1 \frac{f_2}{f_1} \quad (12)$$

### Brookhaven site ENDF Navigation



## Interpreted ENDF file

CO-60 MAT=558 MF=1 MT=451 Library: ENDF/B-VII.1

See original data:  
[Section](#) [Material](#)  
and [Evaluation Summary](#)

### Descriptive Comments

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27-Co- 60  BNL          EVAL-NOV05 Conversion from ENSDF
/ENSDF/                                     20111222
----ENDF/B-VII.1   Material   558
-----RADIOACTIVE DECAY DATA
-----ENDF-6 FORMAT
***** Begin Description *****
**      ENDF/B-VII.1 RADIOACTIVE DECAY DATA FILE      **
**      Produced at the NNDC from the ENSDF database   **
**      Translated into ENDF format by:                **
**      T.D. Johnson, E.A. McCutchan and A.A. Sonzogni, 2011 **
*****
ENSDF evaluation authors: J. K. TULI
Parent Excitation Energy: 0.0
Parent Spin & Parity: 5+
Parent half-life: 1925.28 D 14
Decay Mode: B-
***** Energy Balance *****
Mean Gamma Energy: 2.504E3 +- 3.522E-1 keV
Mean X-Ray+511 Energy: 7.972E-4 +- 5.301E-5 keV
Mean CE+Auger Energy: 3.630E-1 +- 3.252E-3 keV
Mean B- Energy: 9.641E1 +- 2.419E-1 keV
Mean B+ Energy: 0.000E0 +- 0.000E0 keV
Mean Neutrino Energy: 2.232E2 +- 5.863E-1 keV
Mean Neutron Energy: 0.000E0 +- 0.000E0 keV
Mean Proton Energy: 0.000E0 +- 0.000E0 keV
Mean Alpha Energy: 0.000E0 +- 0.000E0 keV
Mean Recoil Energy: 0.000E0 +- 0.000E0 keV
Sum Mean Energies: 2.824E3 +- 7.255E-1 keV
Q effective: 2.824E3 keV
Missing Energy: 1.333E-1 keV
Deviation: 4.720E-3 %
***** End Description *****
```

Page generated by program 'endf' written by R.E.MacFarlane, 8 April 1998, [rjxm@lanl.gov](mailto:rjxm@lanl.gov)

Figure 10: Clicking on “Info” as shown in Figure 9 leads here, where the data is provided.